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Assessment of assembly homogenized two-steps core dynamic calculations using direct whole core transport solutions

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ABSTRACT

The impact of the approximations in the "two-steps" procedure used in the current generation of nodal simulators for core transient calculations is assessed by using a higher order solution obtained from a direct, whole core, transient transport calculation. A control rod ejection accident in an idealized minicore is analyzed with PARCS, which uses the two-steps procedure and DeCART which provides the higher order solution. DeCART is used as lattice code to provide the homogenized cross sections and kinetics parameters to PARCS. The approximations made by using (1) the homogenized few-group cross sections and kinetic parameters generated at the assembly level, (2) an effective delayed neutrons fraction, (3) an effective fuel temperature and (4) the few-group formulation are investigated in terms of global and local core power behavior. The results presented in the paper show that the current two-steps procedure produces sufficiently accurate transient results with respect to the direct whole core calculation solution, provided that its parameters are carefully generated using the prescriptions described in the present article.

1. Introduction

In order to perform three dimensional transient calculations in conventional nuclear power reactors, two broad classes of methods have been developed. Space–time factorization schemes, e.g. quasistatic and point-kinetics techniques (Ott and Neuhold, 1985), provide an efficient way to solve the time dependent Boltzmann equation at the cost of reduced accuracy. The other class of methods uses a finite-difference approximation (with respect to time) to the time-dependent neutron transport equation which provides an accurate solution but for a higher computational cost. The current industry practice for power reactors relies on the latter class of methods. It employs the so called "two-steps" procedure that involves (1) pregeneration of assembly-homogenized few-group cross sections (XS) and discontinuity factors (DF), which are collectively called group constants; and uses (2) diffusion theory-based time-dependent core calculation. Errors are introduced in the two-steps transient calculation primarily by the use of pregenerated few-group assembly homogenized XS, the incorporation of thermal feedbacks, and the modeling of the lower energy of the delayed neutrons.

The goal of the present article is to assess and quantify the impacts of those approximations by using a higher-fidelity solution obtained from a direct, whole core, transient transport calculation. The idea is to evaluate the accuracy of the simplified model (two-steps procedure) by a numerical comparison with a more complex model (direct, whole core transport calculation) for a control rod ejection accident in an idealized minicore. The two-steps procedure is performed by using the PARCS code (Joo et al., 1998) which is one of the typical nodal core simulators used for Light Water Reactor analysis. The direct whole core solution is provided by the DeCART code (Joo et al., 2004). It should be noted that the conclusions drawn from the present study are also applicable to other code systems using the two-steps approach like CASMO-4/SIMULATE-3.

PARCS has been verified against the results of other similar codes (Joo et al., 1996; Kozlowski et al., 2004). The validation of PARCS has been done against a limited set of measured data: a turbine trip event at the Peach Bottom boiling water reactor (PBTT) (Lee et al., 2002), a Boiling Water Reactor stability event (Kozlowski et al., 2014) and a control rod ejection event in the Special Power Excursion Reactor Test (SPERT) Wang et al., 2013.

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DeCART is capable of direct whole core steady state and transient calculations. It employs the planar method of characteristics (MOC) solution based three-dimensional (3-D) coarse mesh finite difference (CMFD) formulation. Transport theory based solvers can also be used in the axial direction (Hursin et al., 2014). DeCART can perform transport calculations (Cho et al., 2005) retaining sub pin level heterogeneity while using more than 40 energy groups. Since it treats every pin explicitly, the individual pin wise Doppler Effect can be incorporated without the averaging used in nodal codes such as PARCS. The delayed neutrons spectrum is also explicitly represented. But most importantly, the direct whole core calculation in DeCART does not introduce the errors associated with the use of the single assembly spectrum for the group condensation and homogenization of XS. The capability of DeCART to perform full core reactivity insertion accident (RIA) analyses and the comparison of DeCART with the two-steps approach was reported in numerous previous articles (Hursin et al., 2010; Hursin et al., 2012; Hursin et al., 2013). A systematic assessment of the various parameters affecting the two-steps calculation has, however, not been published yet.

Since the most important concern regarding the accuracy of the two-steps procedure is the adequacy of single assembly based few group constants, it was decided to construct the problem so as to minimize the consequences of the errors in the assembly homogenized XS. The goal is to focus on the effect of the kinetics parameters. Therefore, the full core PWR model considered in Hursin et al. (2012) is simplified and reduced to a two-dimensional (2-D) 5×5 fresh assembly core model in which a reflective boundary condition is used in all directions. Radial heterogeneity is introduced by loading fuel assemblies of different enrichments. For consistency, the DeCART is used as a lattice code to generate the few-group constants and kinetic parameters for PARCS.

Finally a word needs to be said about the generation of the thermal hydraulic feedbacks. Even though DeCART has been coupled to Computational Fluid Dynamics codes (Weber et al., 2006; Hursin et al., 2008) for steady state and transient analysis, only simplified thermal–hydraulic solvers are used in this work, since its primary focus is the assessment of the neutronic approximations of the two-steps procedure. The same simplified one-dimensional thermal–hydraulic solver is used in DeCART and PARCS. It solves the 1-D radial heat conduction and 1-D axial heat convection problems. The fundamental difference between PARCS and DeCART solvers is the level at which the thermal hydraulic feedbacks are provided. In DeCART, the thermal–hydraulics equations are solved separately for each fuel rod. Conversely in PARCS, the thermal–hydraulics equations are solved for each fuel assembly and only an assembly averaged fuel temperature is used to determine the Doppler Effect. It was shown in Hursin et al. (2015) that this difference has a small impact on the transient results.

The procedure to generate the homogenized group constants for PARCS with DeCART is described in Section 2, as well as the functionalization scheme used to incorporate the thermal feedbacks in PARCS. The generation of the kinetics parameters is then described in Section 3. The dependence of the steady-state solution on the number of energy groups and the XS generation procedure is presented in Section 4 followed by an assessment of the sensitivity of the transient results to the delayed neutrons fraction and to the Doppler Effect modeling parameters in Section 5. Section 6 concludes the paper.

2. Core model and homogenized parameter generation

The transient event considered is a control rod ejection at Hot Zero Power (HZP) conditions. The core model and the calculation conditions are detailed below. The procedure to generate homogenized XS for PARCS using DeCART is described in this section as well.

2.1. Model core and transient scenario

The model consists of a 5×5 fuel assembly array as shown in Fig. 1(a), in which three types of 17×17 assemblies are loaded. The assembly layout is shown in Fig. 1(b) and the spatial discretization of the fuel in DeCART is given in Fig. 2. The gap between fuel and cladding is explicitly modeled. The values of the modeling parameters of interest are listed in Table 1. Reflective boundary conditions are used axially and radially. The core k_{eff} is set to 1.0 by adjusting the boron concentration in the moderator. The initial thermal–hydraulic conditions are specified in Table 2.

The transient is initiated by the instantaneous ejection of a control rod located at the periphery of the core (dotted circle in Fig. 1(a)). The rod worth is set to 1.21$ by adjusting its B_4C concentration. It is expected that the core power will rise rapidly and then decrease to an asymptotic power level for which the negative reactivity caused by the elevated fuel temperatures compensates the positive reactivity inserted by the withdrawal of the control rod.

2.2. Group constant generation with DeCART

As reported in previous work (Hursin et al., 2012), the capability to generate the group constant data needed by PARCS at the assembly level was implemented in DeCART. For this work, the capability to generate the assembly homogenized group constants at the core level was added. Both the generation methods and procedure are briefly explained below.

2.2.1. Group constant generation based on single assembly calculation

The infinite medium spectrum is used in the energy condensation of the XS instead of the critical spectrum. Even if it is not part of the typical two-steps procedure, such choice will not change the outcome of the present study since the goal is to perform a code-to-code comparison. However, as a consequence, the diffusion coefficients are obtained by the inverse of the transport XS, and not as a by-product of the critical spectrum calculation as in Hursin et al. (2012).

The core model shown in Fig. 1(a) requires the generation of three sets of macroscopic XS libraries: one for each fuel type A, B and C. The group condensation is performed in three group structures to result in 2G, 4G and 8G XS data. The energy bounds of the various group structures are coming from Downar and Kozlowski (2006) and are designed for transient analysis. The lower bounds (LBs) of the three group structures are given in Table 3. The effect of the number of energy groups on the two-steps procedure is analyzed for steady state and transient calculations in Sections 4.2 and 5.1.3.

In PARCS, the functionalization scheme used to represent the macroscopic XS is assumed to be piecewise linear with respect to the changes in control rod (CR) insertion, boron concentration (B), moderator density (ρ_M),[2] and fuel temperature (T_F). In order to generate the XS derivatives, a set of branch calculations are performed by changing sequentially the corresponding state variable with respect to the reference condition. The state parameters of the reference and branch calculations are coming from Downar and Kozlowski (2006) and are summarized in Table 4. It should be noted that the perturbation of the various state parameters are fairly large. Consequently, cross section interpolation error cannot be excluded from the overall cross section generation process. Such

[2] When the moderator density is perturbed, the number density of H and O are changed. The moderator temperature is modified accordingly.